NON-PUBLIC?: N
ACCESSION #: 9303050080
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Oconee Nuclear Station, Unit 3 PAGE: 1 OF 11

DOCKET NUMBER: 05000287

TITLE: Reactor Trip On Lost Signal Due To Technician Inattention
To Details, Followed By Operator Misalignment Of
Automatic Emergency Feedwater Paths
EVENT DATE: 01/26/93 LER #: 93-01-00 REPORT DATE: 02/25/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: S.G. Benesole, Safety Review TELEPHONE: (803) 885-3518
Manager

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED:

ABSTRACT:

On January 26, 1993, at 1005 hours, while operating at 100% Full Power, Oconee Unit 3 tripped from a Reactor Protective System anticipatory trip signal. While troubleshooting a problem in Unit 3's Power Factor meter transducer, Instrumentation and Electrical technicians incorrectly tested the voltage input of the transducer with a multimeter in the current measuring mode. This resulted in a partial loss of power to the Generator Output Megawatt meter and a false signal to the Integrated Control System. The Turbine control Valves opened in response to this false signal to recover the apparent lost megawatts. A large decrease in Feedwater Pump discharge pressure caused an Anticipated Transient Without Scram Mitigation System Actuation Circuitry actuation, starting the Emergency Feedwater pumps and tripping the Main Turbine. Main Turbine Anticipatory trip signal tripped the reactor. Post Trip response was normal. During the trip recovery, while transferring from the Emergency Feedwater to the Main Feedwater pumps, a loss of automatic initiation of both emergency feedwater flow paths resulted when both Emergency Feedwater Control Valves were not placed in "AUTO" as directed by procedure. The cause of the unit trip was Inappropriate Action (Improper Action, Lack of Attention to Detail). The cause of the second event was Inappropriate Action (Improperly Followed the Correct Procedure). Corrective actions included replacement of the blown fuses, faulty transducer, and revision of the station drawings and individual counseling to improve personnel performance.

END OF ABSTRACT

BACKGROUND

The Integrated Control System (ICS)EIIS:JA! provides fully automatic control of reactor power, steam generation rate, and generated load by processing selected signals of measured plant parameters. The system coordinates the thermal power of the Reactor Coolant System (RCS)EIIS:AB! with the heat removal capability of the Steam Generators (SG) during both steady state and transient operating conditions. Within the ICS, the unit megawatt (MWe) load demand signal is compared to the actual generated MWe to produce a MWe error. This error signal is applied as a bias to the main steam header pressure setpoint, modifying the pressure setpoint for the turbine controls.

The Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) is divided into separate control logic and output groups. The AMSAC system has a minimum of two channels with two out-of-two logic. The system is actuated when both Main Feedwater (FDW)EIIS:SJ! Pumps Turbine Control Oil Pressure is low or both FDW Pumps discharge pressure is low. AMSAC is designed to mitigate the effects of a transient associated with a lose of Main FDW. During the loss of FDW, AMSAC will automatically start the Emergency Feedwater (EFDW)EIIS:BA! pumps and trip the Main Turbine to prevent a serious RCS over pressurization, maintain fuel integrity and protect against exceeding 10CFR100 release requirements.

The Reactor Protective System (RPS)EIIS:JC! monitors parameters related to the safe operation of the plant and protects the against fuel clad damage and the RCS against damage caused by high system pressure. There are four RPS channels, it takes actuation of at least two of these channels to produce a reactor trip signal. Thus, two out-of-four logic is produced. The generated trip signal will open all Control Rod Drive breakers. Two anticipatory trip signals will actuate the RPS. One is the Main Turbine Trip Anticipatory Trip and the other is both FDW Pumps Trip Anticipatory Trip. Each of these will produce an RPS signal to trip the reactor if proper conditions exist.

The Main FDW system takes a suction from the Condensate EIIS:SD! system, further preheats the feedwater in the feedwater heaters and delivers feedwater to the SGs with two steam driven FDW pumps. ICS controls the amount of flow by throttling two sets of control valves, one for each SG, and FDW pump speed to ensure pump discharge pressure is sufficient to force water into the SGs.

The EFDW System is designed to start automatically upon the loss of FDW. Both Motor Driven Emergency Feedwater (MDEFDW) pumps and the Turbine Driven Emergency Feedwater (TDEFDW) pump will automatically start on a lose of both FDW pumps or low FDW header pressure. SG levels will be controlled

automatically by the EFDW Control Valves (FDW-315 and 316). All EFDW automatic initiation logic and control features are independent of the ICS.

The Power Factor Transducer produces a DC output signal that provides an indication of Power Factor in the Control Room. The power source for the transducer is a fused three-phase circuit from the output of the Turbine Generator through a metering potential transformer. This power source also supplies the transducer that produces the Generator Megawatt output signal fed to the ICS.

EVENT DESCRIPTION

On September 28, 1992, Oconee Unit 3 returned to power operations from a refueling outage. A work request was initiated because Unit 3's Power Factor (PF) meter did not respond properly.

On October 6, 1992, the Unit 3 Reactor Coolant System (RCS) was cooled down and depressurized to 220 degrees F/290# for Control Rod Drive Stator and Position Indication Tube replacement. At approximately 1330 hours, an Instrumentation and Electrical (I&E) technician (Electrician) and an I&E technician (Relay) began troubleshooting the problem with Unit 3's PF meter. They found that the current input to the PF Transducer (PFT) was wired incorrectly. This was determined by referencing Oconee Drawing O-2792-A. Next, they verified that the PF meter was functioning properly, then, wired the PFT according to the drawing. It was decided to wait until the Turbine Generator EIIS:TB! was on-line to check response.

October 14, 1992, 0315 hours, the Turbine Generator was placed on line. The PF meter was not indicating properly. work on this equipment was delayed due to other priority work.

On January 26, 1993, at 0905 hours, the Unit 3 Control Room Senior Reactor Operator gave clearance to resume work on the PFT. At 0917 hours, I&E technicians (Relay) "A" and "B" both verified that they were on the correct unit and component. I&E technicians "A" and "B" began their investigation of the PFT problem by attempting to verify that the appropriate voltage and current conditions existed to and from the transducer. They found that the phase to phase input AC voltage was 192 volts, 57 volts to 97 volts higher than that indicated on the transducer reference label. They proceeded to check the phase to neutral input AC voltages and found all three phases were correctly indicating 111 volts. A review of oconee drawings and printed wiring diagram on the PFT was initiated by the I&E technicians. The two Oconee drawings disagreed with the wiring diagram on the transducer. The I&E technicians changed the wiring on the PFT to match the manufacturer's wiring diagram for the higher voltage applied to the

transducer. Again, measuring the phase to phase input AC voltage, the I&E technicians found the same voltage as before. Measurements were taken for DC output current and none was found. Input AC current reading was made with the Fluke Model 77 multimeter and found acceptable at 3.8 amps. The I&E technicians concluded that the PFT was faulty and the Oconee drawings were not correct. At 0956 hours, the Fluke multimeter was removed and the wiring was returned to its original positions. As a double check, the I&E technicians attempted to check the phase to phase input AC voltage. The Fluke multimeter read zero. The I&E technicians immediately realized that the Fluke meter was still aligned to read current and corrected the meter to read voltage. Continuing with their voltage readings they found that the phase to phase input AC voltage still read zero. They concluded that the Fluke multimeter was damaged. Unknown to the I&E technicians, the metering potential transformer's "X" and "Y" phase fuses had blown because of the improper multimeter alignment.

During the investigation of the PFT problems, operations and I&E personnel were involved with an Instrument Air (IA) system leak in the Penetration Room. An IA isolation was in progress to allow repair of the leak. At approximately 1003 hours, the Control Room personnel observed a system transient in progress and believed it was associated with the IA isolation and operators were dispatched to return the IA system to service.

At 1004:07 hours, a large MWe error statalarm was received. Integrated Control system (ICS) increased output signals to near maximum: Steam Generator Reactor Demand, Reactor Demand and Feedwater Demand. Operations also observed that Main Steam pressure was decreasing. At 1004:11 hours, reactor power had increased to approximately 101% and was held there by the ICS high power limiter. Due to a main steam pressure error signal (50 psi error from setpoint), the ICS Turbine Master station tripped to manual, ICS went into "track" and Steam Generator Reactor, Reactor and Feedwater demands followed the decreasing MWe signal. The Reactor Operators (RO) observed that indicated electrical output made a step decrease from approximately 887 to 300 MWe and that the Main Turbine Control Valves had travelled to full open.

At 1004:27 hours, main steam pressure had reached its low pressure alarm and continued to decrease. RO "A" attempted to close the Turbine Control Valves by decreasing the ICS Turbine Master signal manually. Seeing no immediate main steam header pressure response, the RO stopped his attempt. In parallel to this action, ROs attempted to locate a possible main steam line break and none was found. At this point, the Unit 3 Supervisor, suspecting that the IA work could be the cause, instructed the I&E crew to return the IA system to an operating status. The Operators delayed further actions thinking that this would return the plant to normal status; however, it did not. Operations Unit 3 Supervisor and the ROs continued to

monitor the plant and discussed the possibility of manually tripping the reactor.

At 1006:35 hours, Main Feedwater pump (FDW) Pump discharge pressure low statalarm was received.

At 1007:54 hours, the Anticipated Transient Without Scram (ATWS) Mitigation System Actuation circuitry (AMSAC) Channel 2 actuated. RO "A", observed that main steam pressure had decreased to approximately 500 psi and attempted again to close the Turbine Control Valves by use of the ICS Turbine Master station. By 1007:57 hours, AMSAC channel 2 actuation had cleared.

At 1008:08 hours, AMSAC Channel 2 actuated, again. At 1008:14 hours, AMSAC Channel 1 actuated. With two channels of AMSAC actuated the following automatic actions occurred: at 1008:14 hours, 3A Motor Driven Emergency Feedwater Pump and the Turbine Driven Emergency Feedwater Pump started, at 1008:20 hours, 3B Motor Driven Emergency Feedwater Pumps started, and the Main Turbine tripped. At 1008:23 hours, the reactor tripped because of an anticipatory trip signal from the Reactor Protective System. At 1008:16 hours, AMSAC Channels 1 and 2 actuation cleared. The Operators took manual actions as directed by the Emergency operating Procedure (EP/3/A/1800/01).

By 1009 hours, Main Steam and Main Feedwater discharge pressures had recovered. After the reactor tripped, the Feedwater pumps responded in automatic and increased their discharge pressure, such that, the Turbine Driven Emergency Feedwater pump shutdown. At 1009:57 hours, Main Feedwater discharge pressure decreased and AMSAC channels 1 and 2 actuated again, restarting the Turbine Driven Emergency Feedwater pump.

By direction of the Unit 3 Supervisor, 3A and 3B Main Feedwater Pumps were manually tripped by RO "B" at 1010 hours. 3A and 3B Motor Driven Emergency Feedwater pumps continued to supply feedwater to the Steam generators. The Turbine Driven Emergency Feedwater pump was locked out as directed by the Loss of Main Feedwater Procedure (AP/3/A/1700/19).

I&E technician "B" contacted the Component Engineer to obtain a replacement transducer for the PFT and informed him of the wiring and drawing discrepancies. I&E technician "A" called his supervisor to inform him of the problems encountered and said that they may have been the cause of a unit trip.

Specific post-trip parameters remained within the trip envelope and acceptable limits. Control Rod drive breakers tripped and all Control Rods inserted into the core. Reactor Coolant System (RCS) pressure increased to 2166 psig, then decreased to 1917 psig and controlled at approximately 2155

psig. Pressurizer inventory remained on scale between a high of 223 inches at the time of the trip to a low of 102 inches. RCS temperature converged smoothly to approximately 555 degrees F. Steam Generator pressure reached a low of 595 psig pre-trip to a high of 1040 psig post-trip, then controlled at approximately 1025 psig. Main Steam Relief Valves reseated within minimum reseat pressures.

An investigation began to determine the cause of the unit trip. At approximately 1200 hours, the investigation confirmed that the work on the PFT caused the plant transient and resulting unit trip.

At 1229 hours, Main Feedwater was reestablished per OP/3/A/1106/02 and AP/3/A/1700/19 (Lose of Main Feedwater). Both Motor Driven Emergency Feedwater Pumps were shutdown and their controls placed in "AUTO". Also, the Turbine Driven Emergency Feedwater pump switch was placed in "AUTO". With RO "B" reading AP/3/A/1700/19 (Loss of Main Feedwater), enclosure 6.5 (Reestablishing Main Feedwater) and RO "A" performing the steps, Main Feedwater was re-established. 3FDW-315 and 3FDW-316 (SG '3A' EFDW Control Valve and SG '3B' EFDW Control Valve) were placed in manual and shut per the AP. Later steps of Enclosure 6.5 direct the operator to place the controls for 3FDW-315 and 3FDW-316 and four other feedwater control valves in "AUTO". The controls for 3FDW-315 and 3FDW-316 were not placed in "AUTO" as directed.

At 1805 hours, RO "C" discovered 3FDW-315 and 316 in manual while performing a control board review. With 3FDW-315 and 3FDW-316 in manual, both emergency feedwater flow paths are not automatically available. Technical Specifications state that if both emergency feedwater flow paths are inoperable, immediately initiate corrective action to restore at least one associated emergency feedwater flow path to operable status. Immediate corrective actions were not taken. This was a violation of Technical Specifications 3.4.1 from 1229 hours to 1805 hours. 3FDW-315 and 316 were placed in automatic returning the system to operable status. At 2125 hours, a four hour non-emergency red phone notification was made to the NRC.

The reactor was returned to criticality at 2348 hours.

Exempt Change # OE-5509 was initiated to correct the wiring and install a new Power Factor transducer. This Exempt Change also corrected the electrical system drawings, O-2707 and O-2792.

Unit 3 was returned to power operations on January 27, 1993, at 0330 hours.

CONCLUSIONS

The initiating cause of this event is Inappropriate Action (Improper Action, Lack of Attention to Details). The I&E technicians (Relay) had set the Fluke multimeter to test for a current measurement. With the meter in this alignment, the meter becomes a part of the circuitry. To measure for a voltage potential in this alignment would create a short in the circuit. In this event, the shorted circuitry caused the 6 amp fuses in the "X" and the "Y" phases of the metering potential transformer to blow. The Fluke multimeter has an installed 10 amp fuse that is designed to protect test equipment from this type of event. It did not blow since the 6 amp fuses opened first. The Stop Think Act Review (STAR) work practice is a positive method of accomplishing that task. Had the I&E technicians performed the self-checking technique, Stop Think Act Review (STAR) work practice, the mispositioned Fluke multimeter leads could have been noticed before testing the circuitry and prevented the unit trip.

Oconee Nuclear Station had within the past two years one incident involving the incorrect use of a multimeter documented in LER 287/92-01. On January 14, 1992, I&E technicians were performing trouble checks on a suspected faulty controller in the ICS feedwater control circuits. The I&E technicians used an instrument with the test leads configured for current measurement rather than voltage, causing a false signal to be introduced into the controller resulting in a unit trip. Corrective actions included the use of blank plugs installed in the current measuring jacks of the Fluke 8600 multimeters upon issue. As a result, communication was issued to all three nuclear sites to increase the awareness of the importance of attention to detail. I&E continuing training completed May 1992 did cover LER 287/92-01 and the self verification process; however, the corrective action, concerning the multimeters, was inadequate due to its limited scope of not including all multimeters. These corrective actions have been ineffective in preventing this event. This incident is considered recurring. The failure of the Scientific Columbus, Power Factor Transducer, Model PF-34A4 is not NPRDS reportable.

For this event, Station management and Component and System Engineering management plans to reinforce the "STAR" work practice by special emphasis on self-checking. The practice of placing plugs in the current jacks of multimeters will be extended to all multimeters.

The root cause of the second event is Inappropriate Action (Failure to Follow Procedures, Improperly Followed the Correct Procedure). While reestablishing Main Feedwater, the Control Room Operator failed to place the controls for 3FDW-315 and 3FDW-316 (SG '3A' EPDW Control Valve and SG '3B' EFDW Control Valve) in "AUTO" as directed by AP/3/A/1700/19, enclosure 6.5, step 1.12. Immediate actions were not taken to return the emergency feedwater flow paths to service as directed by Technical Specifications. This resulted in a violation of Technical Specification 3.4.1 which states

that all three emergency feedwater pumps, both flow paths and associated initiation circuitry must be operable. Enclosure 6.5, Step 1.12, which directs the operator to place the controls in "AUTO", may be performed in any order. RO "B" said that he had been reading each step deliberately. The first of the six valves listed in step 1.12 are 3FDW-315 and 3FDW-316. RO "B" verified that four of the six valves were in "AUTO", but could not remember if he had read the step or RO "A" did not hear the communication. At the time of this occurrence, RO "B" was working concurrently through the feedwater start up procedure and the abnormal procedure and did not verify the completion of step 1.12. The ROs experienced no major distractions during this evolution other than some nuisance Radiation Area monitor and Auxiliary steam pressure statalarms.

Also, contributing to the second event was the failure to clearly communicate operations management's expectations concerning the use of place keeping aide. Provided within the Abnormal Procedures are check-off blanks as place keeping aids that should be checked or initialed as the steps are completed. This aid was not used. Also, it was noted during operator simulator training, the use of place keeping aids was not emphasized by the training instructors. 3FDW-315 and 3FDW-316 controls were found out of position during a control board review and placed in "AUTO". No equipment failure was associated with this part of the event and is not NPRDS reportable.

A review of past Problem Investigation Reports (PIR) indicates one similar event occurred in 1992 (Reference LER 270/92-03). During a startup of Unit 2, following refueling outage in February of 1992, Technical Specifications defined Low Temperature Overpressure Protection requirements were violated due to the High Pressure Injection (HPI) system being activated. An HPI pump breaker was racked in and the discharge valves were not de-activated. One root cause of this event was Inappropriate Action (Improperly following the correct procedure). The operator did not determine the necessary actions to be performed, even through the procedure specifically outlines the system requirements. Thus, the second part of this event is considered recurring.

The following equipment failure occurred during this event, but was not a causal factor of the event. 3AS-10 (Steam Seal Regulator Auxiliary Supply Regulator) failed to control Steam Seal pressure during the swap over from Main Steam. The delayed response caused a slight reduction of Main Condenser vacuum to approximately 24 inches. Manual operation by a RO was necessary to regain Steam Seal header pressure. Three times, 3AS-10 has failed to function properly during plant operations. It is currently under evaluation. This type of valve is used on all three Oconee units. The valve is a Fisher-Governor, Model 657, control valve. This failure is not NPRDS reportable.

There were no personnel injuries, radiation exposures, or releases of radioactive materials associated with this event.

CORRECTIVE ACTIONS

Immediate

1. Operations personnel took the appropriate actions per the Emergency Operating Procedure to bring the unit to a stable hot shutdown condition.

Subsequent

1. The Control Room Operator noted and corrected the Steam Seal system problem (3AS-10).

2. Instrumentation and Electrical technicians located the problem with the Power Factor meter circuitry. All three fuses and the Power Factor transducer were replaced.

3. An Exempt Change was implemented to change the wiring on the transducer and to revise the system drawings.

4. 3FWD-315 and 316 switches were placed automatic.

5. Operations and Instrumentation and Electrical personnel directly involved with this event have been counseled concerning their actions to improve personnel performance.

Planned

1. Instrumentation and Electrical management will expand the policy to having blank plugs installed in the current measuring jacks of all multimeters.

2. Station management and Component and System Engineering management will place additional management emphasis on the "STAR" self checking technique and stress the individuals accountability for making it a routine part of every action.

3. Operations management will emphasize the use of place keeping aide in the Emergency Operating Procedure and Abnormal Procedures during actual emergencies and during simulator training.

4. Operations and Instrumentation and Electrical management will discuss this incident with all appropriate personnel.

5. Operations management will review operations Management Procedure 1-9 to ensure expectations are adequately reflected concerning the use of place keeping aids.

6. Engineering will review the metering potential transformer and Integrated Control System transducer circuitry for possible modification of the transducer voltage input circuit to provide alternate power source with automatic transfer or automatic alarmed signal blocking. (i.e., voltage balance relaying 60!).

7. The generic control problems associated with AS-16 on all three units will be investigated to determine failure mode and to initiate corrective actions.

SAFETY ANALYSIS

The reactor tripped on an anticipatory trip signal from the Reactor Protective System. All accident mitigation systems and their support systems were available and performed as designed. All full length control rods dropped into the core and the reactor was shutdown and maintained in a safe shutdown condition.

The pressurizer safety valves and the power operated relief valve were not actuated. There were no Engineering Safety Features Actuation System initiation. There was no lose of Reactor Coolant System water inventory.

Post trip response was normal for the level of reactor power. This includes Pressurizer level, Reactor Coolant System pressure, temperature and flow, and Steam Generator level and pressures. Control Room personnel actions safely controlled the reactor and maintained it in a safe shutdown condition.

The Reactor Protective System anticipatory reactor trips are a conservative non-nuclear trip function used to generate a reactor trip before the actuation of the Reactor Protective System nuclear trips. These trips limit the extent of overheating of the reactor Coolant System that could occur during a turbine trip or a total loss of main feedwater.

Incorrectly testing voltage output of the transducer with a multimeter in the current measuring mode resulted in a short, two blown fuses in the transducer power input circuitry and a partial loss of power to Generator Output Megawatt meter. The reduced generator megawatt output signal fed to

the Integrated Control System responded as if an actual lose of megawatts had occurred. The Anticipated Transient Without Scram Mitigation System Actuation Circuitry actuation started the Emergency Feedwater pumps and tripped the Main Turbine. The Main Turbine Anticipatory trip signal tripped the reactor. These protective circuits are designed to protect the Reactor Coolant System from an overpressure condition and prevent challenging the Power Operated Relief Valves. These protective systems operated as designed, thus mitigating the consequences of this event.

Operability of the Emergency Feedwater system assures the capability to remove decay heat before removal via the Decay Heat Removal system, in the event that the Main Feedwater system is inoperable. In the second event, where both 3FDW-315 and 316 were not placed in "AUTO", both Motor Driven Emergency Feedwater pumps and Turbine Driven Emergency Feedwater pump controls were in "AUTO" and started automatically. The Control Room operator has sufficient Control Room indications of Steam Generator level and pressure and would immediately be aware of any degraded situation upon which to take the appropriate manual actions (such as, manually controlling of 3FDW-315 and 316) to maintain Steam Generator inventory. Also, a sufficient depth of backup measures is provided to allow the Steam Generator water inventory to be maintained by any one of diverse methods (i.e., Hotwell and Condensate Booster Pumps, supply from another Unit's emergency feedwater system, Auxiliary Service Water system or Standby Shutdown Facility Auxiliary Service Water system). During an event where Emergency Feedwater would be required, AP/3/A/1700/19 (Lose of Feedwater) gives specific directions to the operator to ensure that adequate feedwater flow is available by manually controlling 3FDW-315 and 316.

There were no releases of radioactive material, radiation exposures, or personnel injuries associated with this event. The health and safety of the public was not affected by this event.

ATTACHMENT 1 TO 9303050080 PAGE 1 OF 1

Duke Power Company
Oconee Nuclear Generation Department J.W. HAMPTON
P.O. Box 1439 Vice President
Seneca, SC 29679 (803) 885-3499 Office

(803) 373-5222 FAX

DUKE POWER

February 25, 1993

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Site
Docket Nos. 50-269, -270, -287
LER 287/93-01

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 287/93-01, concerning a partial loss of generator output signal which caused a reactor trip and a loss of automatic initiation of both emergency feedwater flow paths which occurred during unit recovery.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

J. W. Hampton
Vice President

/ftr

Attachment

xc: Mr. S. D. Ebneter INPO Records Center
Regional Administrator, Region II Suite 1500
U.S. Nuclear Regulatory Commission 1100 Circle 75 Parkway
101 Marietta St., NW, Suite 2900 Atlanta, Georgia 30323

Mr. L. A. Wiens Mr. P. E. Harmon
Office of Nuclear Reactor Regulation NRC Resident Inspector
U.S. Nuclear Regulatory Commission Oconee Nuclear Site
Washington, DC 20555

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